4 RISK INSIGHTS RELATED TO POSTCLOSURE PERFORMANCE ASSESSMENT MODEL ABSTRACTIONS

The system description and the risk insights presented in this report are intended to assist the staff in their prelicensing interactions with the U.S. Department of Energy (DOE) and in reviewing any license application DOE may submit. The staff have not made any determinations regarding the technical conditions or the adequacy of a potential repository at Yucca Mountain at this time. If DOE submits a license application for a repository at Yucca Mountain, the staff will review the information provided by DOE, and make its determinations based on information available at that time.

4.1 Risk Insights on Geological Disposal

Geologic disposal has been internationally adopted as an appropriate method for ensuring protection of public health and safety for very long time periods (e.g., 10,000 years) because deep geologic disposal: (i) limits the potential for humans to come into direct contact with the waste; (ii) isolates the waste from a variety of natural, disruptive processes and events occurring on the surface of the earth; and (iii) limits the transport of radionuclides, after release to ground water, by the natural hydrological and chemical properties of geologic strata comprising a potential repository site. Additionally, it has been widely accepted that a geologic repository is to be comprised of multiple barriers as a means of providing defense-in-depth. The multiple barrier approach includes consideration of both natural barriers (e.g., hydrological properties of rock and soil units, geochemical retardation) and engineered, or human-induced, barriers (e.g., waste package, waste form) as a means to contain and isolate waste.

Understanding the potential risk of high-level waste begins by considering the radionuclides that comprise high-level waste, radionuclides that vary significantly with respect to inventory, radioactive half-life, and radiotoxicity. Table 4-1 provides information on the radionuclides relevant for evaluating releases in the ground water pathway (i.e., radionuclides or daughters of radionuclides with radioactive half-lives of at least 100 years, and sufficient inventory, such that a portion of these radionuclides might be transported to the compliance location via ground water). As shown in Table 4-1, the overall radionuclide activity at 1,000 years is dominated by relatively few radionuclides (i.e., Am-241, Pu-240, Pu-239, Am-243, and Tc-99). The potential risk of the overall radionuclide inventory is determined by weighting the inventory of each radionuclide by its dose conversion factor, a measure of a radionuclide radiotoxicity. The potential risk of the inventory is similarly dominated by the same radionuclides, with the exception of Tc-99, which is not as significant because of its low radiotoxicity (i.e., low dose conversion factor).

Although the entire inventory of high-level waste represents a significant risk, if the inventory were quickly released to the biosphere, current performance assessments of a potential repository at Yucca Mountain indicate that the majority (i.e., greater than 99 percent) of the inventory is isolated from humans during the regulatory period and beyond, because of the effectiveness of the engineered barriers and the attributes of the site (i.e., natural barriers). Evaluating the effectiveness of the multiple barriers requires an understanding of both the potential risk of the high-level waste inventory as well as the attributes of the design and site that affect the release and transport of each radionuclide. For example: (i) long-lived waste

Table 4-1. Inventory (Based on the Activity Present at 1,000 Years) and Weighted Inventory (Based on the Activity Present at 1,000 Years Weighted by the Dose Conversion Factor) of Radionuclides Evaluated in Ground Water Releases				
Radionuclide	**Half-Life, Years**	**Inventory at 1,000 Years, Percent of Total**	**Ground Water Dose Conversion Factor, mrem/yr/pCi/l**	**Weighted Inventory at 1,000 years, Percent of Total**
Am-241	430	54	4.9	56
Pu-240	6,500	25	4.7	25
Pu-239	24,000	18	4.7	18
Am-243	7,400	1.2	4.9	1.2
Tc-99	210,000	0.73	0.0022	0.00033
U-234	240,000	0.13	0.38	0.010
Ni-59	76,000	0.12	0.00032	0.0000083
C-14	5,700	0.065	0.0035	0.000048
Np-237	2,100,000	0.064	6.0	0.080
Nb-94	20,000	0.042	0.0096	0.000052
Cs-135	2,300,000	0.027	0.012	0.000065
Se-79	65,000	0.023	0.013	0.000063
U-238	4,500,000,000	0.016	0.35	0.0012
Cm-246	4,700	0.0032	4.9	0.0033
I-129	16,000,000	0.0018	0.43	0.00016
Th-230	77,000	0.0011	0.74	0.00017
Cl-36	300,000	0.00058	0.0061	0.00000075
Ra-226	1,600	0.00019	1.8	0.000074
Pb-210	22	0.00019	7.3	0.00030

packages are expected to retain their integrity during the period of the highest thermal output of the waste, when the waste-form behavior is most uncertain; (ii) radionuclides are expected to be released slowly from the engineered barrier system once the waste packages are breached; and (iii) radionuclides are expected to travel slowly from the engineered barrier system to the area where potential exposures might occur because of the sorptive properties of the surrounding rock. Thus, multiple barriers, as a defense-in-depth approach, result in a robust repository system that is more tolerant of failures and external challenges.

The risk insights for geologic disposal are developed by understanding the significance to waste isolation of the long-lived waste package, release rates of radionuclides, and transport of radionuclides, in the context of the effect on risk estimates. One approach for understanding and communicating the waste isolation capability of attributes of geologic disposal is to evaluate the releases of radionuclides at certain well-defined locations, such as from the waste package and geologic setting (the potential receptor location or compliance location). This helps characterize the behavior of specific barriers or subsystems of the overall repository. Figures 4-1 and 4-2 represent the effective activity released from the waste package and geologic setting, respectively. Effective activity is determined by weighting the activity for each radionuclide by its dose conversion factor, which allows the releases of radionuclides to be compared on a similar radiotoxicity basis. For the radionuclides shown in Figures 4-1 and 4-2, the majority of radionuclides that exit the waste package are not released from the geologic setting (i.e., at the compliance location) before 10,000 years. The radionuclides that tend to chemically sorb onto rock surfaces (e.g., plutonium, americium, and neptunium) are not estimated to arrive at the compliance location, whereas, radionuclides such as iodine and technetium, which are less likely to chemically sorb, do arrive at the compliance location. The releases of iodine and technetium are barely discernable in Figure 4-2; however, the release rates from the geologic setting are approximately three times smaller than the release rates from the waste package. These figures provide quantitative risk information regarding the magnitude of releases from the waste package, and the attenuation of these releases by the attributes of the geosphere, before they reach the compliance location.

Although the presentation of radionuclide release rates from specific subsystems is useful for understanding specific processes, this type of information does not readily convey the behavior of the spectrum of barriers of the repository system and the collective effectiveness for isolating waste. The staff have developed an approach for representing the waste isolation capabilities of specific attributes of the repository system in the context of the overall system, as a means to enhance understanding and risk insights. This approach represents the following three primary attributes for achieving waste isolation: (i) long-lived waste packages, (ii) slow release of radionuclides from the engineered barriers, and (iii) slow migration of radionuclides in the geosphere. These attributes promote waste isolation by delaying and/or reducing releases of radionuclides to the compliance location. Performance assessment calculations are used to evaluate the effectiveness of individual barriers to isolate waste. For example, delay times are calculated for barriers that principally act to delay the onset of releases or the movement of radionuclides (e.g., waste package lifetime, transport time to move through the geosphere), whereas release rates are calculated for barriers that limit, rather than delay, releases (e.g., solubility limits, limited water contact with waste, spent nuclear fuel degradation rates).

Table 4-2 provides this type of general perspective on the capabilities of the site and design attributes for isolating the radionuclides considered in the ground water pathway. The site and design attributes are divided into three categories affecting waste isolation: (i) delay for the

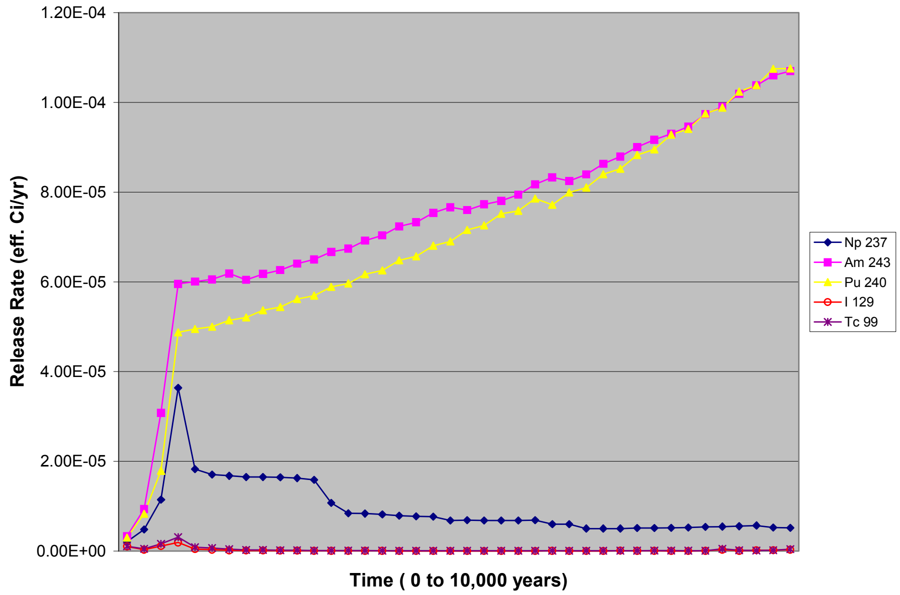

Figure 4-1. Effective Activity Released from the Waste Package. (1 Ci/yr = 3.7 × 10^{10}Bq/yr)

Note: Effective Activity Determined by Weighting Release for Each Radionuclide by its Dose Conversion Factor.

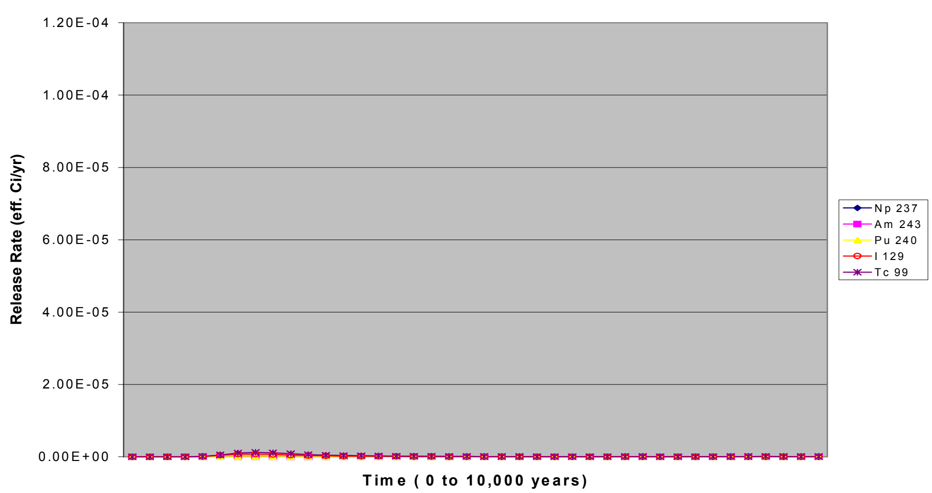

Figure 4-2. Effective Activity Released from the Geologic Setting. (1 Ci/yr = 3.7 × 10^{10}Bq/yr)

Note: Effective Activity Determined by Weighting Release for Each Radionuclide by its Dose Conversion Factor.

Table 4-2. Representation of Effectiveness of the Attributes of Waste Isolation							
	Attributes of Waste Isolation						
	Onset of Release	Release Rate			Geosphere Transport		
Radionuclide	Waste Package	Waste Form	Solubility Limits	Solubility and Limited Water	Unsaturated Zone	Saturated Zone - Tuff	Saturated Zone - Alluvium
Am-241	DDD				DDD	DDD	DDD
Pu-240	DDD			L	DDD	DD	DDD
Pu-239	DDD			L	DDD	DD	DDD
Am-243	DDD			L	DDD	DD	DDD
Tc-99	DDD	LL			D	D	D
U-234	DDD			L	DDD	D	DDD
Ni-59)	DDD	LLL	L	LL	DDD	D	DDD
C-14	DDD	LLL			D	D	D
Np-237	DDD			L	DDD	D	DDD
Nb-94	DDD	LL	LLL	LLL	D	DD	DDD
Cs-135	DDD	LL			DDD	DDD	DDD
Se-79	DDD	LL			DD	D	DD
U-238	DDD	L	LLL	LLL	DDD	D	DDD
Cm-246	DDD	L			D	DD	DDD
I-129	DDD	LL			D	D	D
Th-230	DDD	LL	L	LL	DDD	DD	DDD
Cl-36	D	LL			D	D	D
Ra-226	DDD	LL		L	DDD	DD	DDD
Pb-210	DDD	LL	L	LL	DDD	DD	DDD

Notes: D denotes delay time of at least 10,000 years (DDD); 1,000 years (DD); and 100 years (D).

L denotes limit on release of 10,000 (LLL), 1,000 (LL) and 100 (L) times less than 0.15 mSv (15 mrem).

onset of initial release, (ii) release rates from the engineered barrier system (principally the waste package and waste form), and (iii) transport in the geosphere. The effectiveness of each barrier associated with the attributes of waste isolation is indicated by the letter D or L, which is used to represent three levels of effectiveness by the number of letters present.

When the design or site attribute delays the onset of release or transport in the geosphere, the level of effectiveness was determined according to delays of no less than 10,000 years (DDD), 1,000 years (DD), or 100 years (D). For the release rate, where the attribute of a barrier is not a delay but rather a limitation on the magnitude of the release, the level of effectiveness was determined by whether the magnitude of release, if instantly released to the biosphere, would result in a potential dose of 10,000 (LLL), 1,000 (LL), or 100 (L) times less than 0.15 mSv [15 mrem].

Table 4-2 offers a general explanation for the risk currently estimated for the proposed repository; namely, the variety and number of design and site attributes result in a very limited amount of the high-level waste inventory being transported by ground water to the compliance location. Additionally, from a defense-in-depth perspective, the importance of any one barrier is generally diminished as the number of relatively independent barriers increases. In other words, poor performance of one barrier does not cause a significant increase in the estimated risk; thus, confidence in the overall safety of the repository system is significantly enhanced when there are multiple and effective barriers.

The results presented in Table 4-2 are based primarily on the average behavior of the repository system and provide a useful general overview. However, this approach does not readily lend itself to addressing the uncertainties in estimating the behavior of the repository system. For example, there are uncertainties with mechanical damage of the waste package, and the effect of colloidal transport of radionuclides, that are not directly represented in Table 4 2. The technical details and uncertainties are the subject of the detailed risk insights provided in the remainder of this section.

Additionally, Table 4-2 addresses releases in the ground water pathway and does not address releases to the air pathway from a potential igneous event. Igneous activity has a potential for higher consequences than estimated for the ground water pathway. However, the risk is still estimated to be small from this scenario, because the probability for igneous activity is orders-of-magnitude below the probability for ground water releases. (Sections 4.3.10 and 4.3.11 provide more discussion on igneous activity.)

4.2 Current Understanding of the Postclosure Repository System

This section provides a summary of the staff current understanding of a postclosure repository system at Yucca Mountain. This understanding is based on the process-level and system-level technical information and performance assessment results that are currently available. The staff understanding continues to evolve as information becomes available through the prelicensing activities and interactions.

The system description provided in this section is presented in seven sections:

- Infiltration, percolation, and seepage into the repository
- Degradation of the engineered barrier system, including the waste form

- Radionuclide release from the engineered barrier system
- Flow and transport of radionuclides in the unsaturated zone below the repository
- Flow and transport of radionuclides in the saturated zone
- Biosphere and reasonably maximally exposed individual
- Igneous activity

4.2.1 Infiltration, Percolation, and Seepage

Yucca Mountain has a semiarid climate and currently receives an average of approximately 190 mm [7.5 in] of precipitation per year. Future climate is expected to evolve according to anticipated glacial cycles. Evidence suggests at the last full glacial maximum, average, annual precipitation may have been 1.5 to 2.5 times larger than current climatic conditions, whereas average annual temperatures may have been 5 to 10 °C [9 to 18 °F] cooler. Approximately 95 percent of the precipitation currently falling onto Yucca Mountain is estimated to be removed by runoff, evaporation, and plant transpiration. The remainder infiltrates through the near-surface environment in a heterogeneous spatial pattern and generally percolates vertically downward through the unsaturated tuff toward the proposed repository horizon. However, large-scale features (e.g., fault zones, hydraulic conductivity contrasts at the interfaces between tuff layers) and small-scale features (e.g., variability of hydraulic conductivity within a tuff layer) may complicate the flow paths and cause deep percolation to redistribute spatially. Although surface infiltration is highly episodic, unsaturated flow reaching the repository is generally assumed to be steady and continuous as a result of the damping effect of the permeable and porous rock matrix of the overlying Paintbrush tuff nonwelded horizons.

Deep percolation rates above the repository directly influence water seepage into the drifts and the amount of water entering breached waste packages, which, in turn, facilitates the release of radionuclides from the engineered barrier system into the underlying the repository horizon. The ambient seepage model suggests that only a small fraction of the percolating water will enter the drift by way of dripping from the drift ceiling because of capillary diversion. The effect of the repository thermal pulse also affects seepage into drifts. When ventilation is stopped at the time of closure, the temperature of the wallrock will quickly rise. The quantity of percolating water that will reach the engineered barrier system will be significantly reduced as decaying radioactive waste heats surrounding rock above boiling temperature during the first few thousand years, thereby driving liquid water away. Water evaporated by repository heat will move toward cooler areas where it will condense and may flow back as liquid water toward the drifts (refluxing). A combination of ambient percolating and refluxed water may penetrate back along preferential flow paths, into the thermally perturbed rock that has temperatures above boiling, and seep into the drifts. Not all areas of the drifts will experience above-boiling conditions. Edge-cooling effects allow water to be present in the wallrock throughout the performance period in drift areas near the periphery of the repository.

At early times following repository closure, the drifts likely will remain open and could be effective at diverting water (if water is present) around the drifts by capillary retention in the rock matrix and fracture networks. The drifts, however, may degrade over time and fill with rockfall rubble from the tunnel walls. As a result, an increased fraction of the percolating water may contact engineered barrier components by way of seepage through rock rubble. Drift ceiling irregularities, such as small asperities and lithophysae, may give way to larger irregularities as drift degradation occurs, potentially reducing the amount of water diversion caused by capillary retention.

Film flow along open drift walls is another mechanism for diverting water away from the drip shield and waste package. The formation of a rubble pile in contact with waste packages or shields could provide an additional mechanism for percolating water to come into contact with engineered barriers. During the reflux period, or as the thermal pulse is dissipating, water seepage into the drifts (including along-wall seepage) contributes to the vapor pressure in the drifts and, thus, elevates the relative humidity of air surrounding the engineered components. Movement of water vapor into cooler areas of drifts could produce condensation that results in additional or focused dripping. Water dripping onto engineered components during the thermal period would evaporate and leave a residue, which, along with any dust present, may affect the chemistry of any liquid water that is later present on those engineered components.

The chemistry of water contacting engineered components can strongly affect degradation of those components through aqueous corrosion processes. Estimating the evolution of the near-field environment is complex because of coupled thermal-hydrological-mechanical-chemical processes and changes in the emplacement drift configuration caused by the collapse and rubbling of overlying rocks. Water and gas compositions will be influenced by chemical reactions within the unsaturated fractured rock. Local changes in water and gas chemistry may result from interactions with engineered materials, corrosion products, or both. The presence of rubbled rock will cause higher temperatures within the near-field environment and engineered barrier components. Major processes affecting the evolution of the near-field environment include evaporative processes and mineral dissolution and precipitation, as well as aqueous- and gaseous-phase transport and chemical reactions.

Deep percolation rate also directly influences the transport of radionuclides through the unsaturated zone to the saturated zone. Transport of radionuclides through the unsaturated zone mainly occurs in fractures within the welded units and in the matrix within the nonwelded units. Sorption during matrix flow through vitric and devitrified nonwelded tuff horizons can significantly delay transport of radionuclides to the water table.

4.2.2 Degradation of the Engineered Barrier System

The current DOE design for the engineered components calls for 63,000 metric tons [69,450 tons] of commercial spent nuclear fuel, as well as 7,000 metric tons [7,720 tons] of DOE spent nuclear fuel and solidified high-level waste, to be loaded into waste packages before placement in tunnels cut into the unsaturated tuff approximately 350 m [1,150 ft] below the surface. The commercial spent nuclear fuel generally is in the form of ceramic-like pellets of irradiated uranium-dioxide (UO_2) clad in corrosion-resistant Zircaloy tubes, approximately 0.6 to 0.9 mm [0.024 to 0.035 in] thick. The current waste package design for commercial spent nuclear fuel consists of a 20-mm [0.8-in]-thick Alloy 22 outer container surrounding a 50-mm [2.0-in]-thick Type 316 nuclear-grade stainless steel inner container, to provide structural strength during preclosure operations. The staff understanding is that after the spent nuclear fuel or other high-level waste is loaded, lids will be welded onto the waste packages before they are placed them in the repository; and before permanent closure of the repository, an inverted U-shaped metal drip shield, approximately 15 mm [0.6 in] thick, fabricated from Titanium Grade 7, will be installed over the emplaced waste packages. The bulk of the 7,000 metric tons [7,720 tons] of DOE waste will be in the form of borosilicate glass, poured into stainless steel canisters, and encased in waste packages of similar design to that used for commercial spent nuclear fuel.

The drip shield and waste package can protect the waste form from dripping water while they remain intact, thereby limiting both the timing and magnitude of radionuclide release. The drip shield may also limit the exposure of the waste package to aggressive chemical environments resulting from thermal-hydrological-chemical processes, as well as mitigate mechanical damage to the waste package from falling rocks. These engineered barriers may be compromised by various degradation processes, including corrosion and mechanical damage. The lifetime of the engineered barriers can be influenced by the environmental conditions to which they are exposed; rockfall from drift degradation or seismicity; faulting; or ascending magma from volcanic activity.

The flow of water into a breached waste package will depend on the location and cross-sectional area of the breaches through the waste package. Four simplified categories of failure can facilitate understanding of the performance of the waste package in limiting radiological releases: (i) a limited number of waste packages with small cracks or perforations, (ii) a small number of waste packages with large breaches, (iii) a large number of waste packages with small cracks or perforations, and (iv) a large number of waste packages with large breaches. A limited number of waste package breaches, either large or small, may result from aggressive and highly localized environments, isolated rockfall from drift degradation or seismic events, faulting, and manufacturing defects. Stress corrosion cracking is the main process by which frequent but small cracking of waste packages could occur. The likelihood of stress corrosion cracking can be promoted by aggressive environmental conditions combined with residual stresses resulting from fabrication and closure operations or applied stresses as a consequence of extensive rockfall from widespread drift degradation or seismicity, as well as accidental internal overpressure. Large widespread failures of the waste packages may result from accelerated, localized corrosion because of pervasive aggressive environments or extensive rockfall from large-scale drift degradation or very large seismic events. In this context, the fabrication and closure processes may result in microstructural changes of the container material that can affect significantly the resistance to localized corrosion and mechanical damage, as well as the mode and extent of the resulting failure.

4.2.3 Radionuclide Release from the Engineered Barrier System

The engineered barrier system consists of the waste form, cladding (for spent nuclear fuel) pour container (for vitrified waste), waste package, invert, and drift. Once radionuclides released from the waste package leave the boundary of the drift, they are considered to be in the unsaturated zone, although components of the engineered barrier system, such as the invert, can also be considered to be unsaturated.

The waste form may begin to degrade once the waste package integrity is breached so that air, water vapor, and liquid water can come into contact with it. Although the waste form can degrade in the presence of air and water vapor, the release of most radionuclides of concern from the waste package and through the invert will only occur when liquid water is present to facilitate transport by means of advection (i.e., transport with the flow of water) and diffusion (i.e., transport from areas of high concentration to low concentration by random motion at the molecular level).

Spent nuclear fuel would be the main contributor to the radioactive inventory of the repository. This waste will be in the form of small UO_2 pellets placed in long tubes known as cladding. Most fuel will be clad in zirconium alloy (e.g., Zircaloy), which is highly corrosion-resistant, and will prevent the spent nuclear fuel from coming into contact with air and water as long as it remains intact. A small fraction of cladding will already be failed on placement in the waste packages. In addition, some reactor fuel was clad with aluminum or stainless steel, which has inferior or less predictable corrosion resistance than zirconium. Factors affecting the long-term integrity of the Zircaloy cladding include: (i) localized corrosion and stress corrosion cracking of cladding that is exposed to in-package water; (ii) hydride reorientation and cracking; (iii) creep failure; and (iv) mechanical stresses caused by seismic shaking, rockfall, faulting, and handling during shipment and loading. Under some conditions, small failures of the clad fuel-tubes will allow water and air to cause swelling of the exposed fuel pellets, which can lead to rapid unzipping of the cladding in that tube. Swelling of fuel pellets as well as hydride cracking and creep failure are processes that are likely to have a more pronounced effect on high burnup fuel.

In the case of fuel reprocessing, vitrified glass-like waste will be encased in stainless steel pour canisters before placement in the waste packages. The pour canisters, like cladding, will provide protection beyond what is available from the waste packages alone.

As the waste forms degrade and come into contact with liquid water, radionuclides will be released in the form of dissolved species, particulates, and colloids. The process of radionuclide release from the waste form to liquid water is generally known as dissolution, although it does not necessarily involve dissolved materials only. The water-borne radionuclides may then escape the waste package by advection and diffusion. Dissolution of radionuclides; formation of, or attachment to, mobile colloids; and incorporation into secondary minerals, formed from the degradation products of UO_2, may influence the transport rates and the amounts of radionuclides that are available for transport from the waste package. A small percentage of the waste consists of radionuclides that are very soluble and are expected to be released to the water quickly. The bulk of the radionuclide inventory consists of radionuclides that will be released from the waste form no faster than their solubility limit would allow. Therefore they will not be released from the waste package any faster than the flow of water at the solubility limit would allow. Some solubility-limited radionuclides (e.g., plutonium), particularly those associated with the vitrified high-level waste forms, can form colloids or attach themselves to naturally occurring or human-induced, nonradioactive colloids (e.g., iron oxyhydroxides from corrosion of steel). Association of radionuclides with colloids can increase the effective concentration of the water above the solubility limit of the radionuclide itself. However, colloids behave differently from dissolved constituents, and may be inhibited from transport from the waste package by attaching to internal surfaces (Wilson and Bruton, 1989). In addition, diffusion of colloids is significantly lower than truly dissolved materials because of their much greater size.

Once radionuclides exit the waste package, they must migrate from the waste package through the underlying invert to be released to the unsaturated zone. The current DOE design for the invert consists of a carbon steel support frame backfilled with compacted crushed tuff, up to about 0.5 m [20 in] in thickness, through which the radionuclides must migrate by advection and diffusion to the unsaturated zone. The porous nature of the invert material may delay transport of radionuclides; however, it is necessary to assess the porous flow and sorption properties of the invert material to determine the effectiveness of this barrier.

4.2.4 Flow and Transport of Radionuclides in the Unsaturated Zone below the Repository

The potential repository at Yucca Mountain will be underlain by approximately 300 m [1,000 ft] of unsaturated volcanic rock layers above the water table. The series of unsaturated layers below the repository is comprised of tuffaceous rock exhibiting varying degrees of welding, which affect both the fracture density and matrix conductivity. Densely welded tuffs are brittle and typically develop interconnected fractures, which may allow water to divert around areas of lower conductivity, whereas nonwelded tuffs exhibit low fracture density and higher matrix conductivity.

Dissolved and suspended radionuclides released from the engineered components would be transported by water flowing through the unsaturated tuffs to the water table. Water typically moves vertically downward through the unsaturated tuffs below the repository through a combination of fracture and matrix flow. However, large-scale (e.g., fault zones or hydraulic conductivity contrasts at the interfaces between tuff layers) and small-scale (e.g., the variability of hydraulic conductivity within a tuff layer) features may add complexity to the flow paths. Water tends to move slowly {e.g., currently estimated at 1 m/yr [3.3 ft/yr] and slower} through unsaturated tuff layers when flow occurs predominantly within the rock matrix. As the water flux exceeds the matrix saturated hydraulic conductivity, water will flow through fractures. Water tends to flow more swiftly (e.g., an estimated tens of meters per year and faster) through tuff layers when flow is predominantly through fractures. Current understanding suggests the Calico Hills nonwelded vitric layer is the only unsaturated tuff layer below the repository with sufficient matrix saturated hydraulic conductivity to allow water to flow predominantly within the rock matrix. The thickness of the Calico Hills nonwelded vitric unit layer is spatially uncertain and may pinch-out, resulting in no Calico Hills nonwelded vitric unit layer beneath portions of the repository.

In addition to the advective transport process described above, transport of radionuclides in the unsaturated zone would be affected by molecular diffusion between fractures and the rock matrix, mechanical dispersion, and physico-chemical processes such as sorption and precipitation. Sorption of radionuclides onto mineral surfaces can be a significant retardation mechanism when radionuclides move through the rock matrix because of the large surface area associated with the rock pores; conversely, fracture pathways have relatively limited surface area and thus exhibit limited if any sorption effects. Dissolved radionuclides transported by water within fractures may diffuse from the water within the fractures into the slow-moving water within the rock matrix, thereby limiting the transport of radionuclides in fractures. However, radionuclides transported by fracture flow could have limited time to diffuse from the fractures into the rock matrix because of the high velocity of water in the fractures (i.e., tens of meters per year).

Transport of radionuclides in colloidal form can limit the effectiveness of sorption processes; however, it can be expected that many colloids will be filtered out over long transport paths in geologic systems.

4.2.5 Flow and Transport of Radionuclides in the Saturated Zone

The saturated zone in the vicinity of Yucca Mountain consists of a series of alternating volcanic aquifers and confining units above the regional carbonate aquifer. The volcanic rocks generally thin toward the south and become interspersed with valley fill aquifers to the south and southeast of Yucca Mountain. The valley fill aquifer is composed of alluvium derived from Fortymile Wash, and colluvium from the adjacent highlands to the east and west, as well as lacustrine deposits formed near the southern end of Jackass Flats. The effective porosities of the fractured rock are expected to be lower than the valley fill alluvium, resulting in higher ground water velocities in the fractured tuffaceous rocks.

Dissolved or suspended radionuclides released from the engineered components would be transported by water generally moving vertically downward through the unsaturated tuffs to the saturated zone. Ground water flow, in the saturated zone immediately below the repository, is driven by a small hydraulic gradient, approximately 0.0001, and is directed to the east-southeast through the eastward dipping upper volcanic confining unit and upper volcanic aquifer. Approximately 2 to 4 km [1.2 to 2.5 mi] east-southeast of Yucca Mountain, the hydraulic gradient is larger, approximately 0.001, and ground water is reoriented south through the tuff aquifer. South of Yucca Mountain, approximately 10 to 20 km [6 to 12 mi], radionuclides would enter the highly porous valley fill aquifer.

The transport of radionuclides in the saturated zone would be affected by molecular diffusion between fractures and the rock matrix, mechanical dispersion, as well as physico-chemical processes associated with sorption of radionuclides onto mineral surfaces. Many radionuclides are retarded when moving through the porous alluvium, because of the large surface area associated with porous media. Certain radionuclides, however, such as I-129 and Tc-99, are generally not retarded in geologic systems. Conversely, the fracture paths in the volcanic rock of the saturated zone are characterized by relatively limited surface areas and thus exhibit limited if any sorption effects within the fractures. However, dissolved radionuclides transported by water within fractures may diffuse from the water within the fractures into the slow-moving water within the rock matrix, thereby limiting the transport of radionuclides in fractures. The flow path in the saturated zone is more than 10 times longer than the flow path in the unsaturated zone (i.e., kilometers versus hundreds of meters). Therefore, significantly more time is available for radionuclides to diffuse from the fractures into the rock matrix of the saturated zone.

Transport of radionuclides in colloidal form or attached to colloids can limit the effectiveness of sorption processes; however, it can be expected that many colloids will be filtered out over long transport paths in geologic systems.

4.2.6 The Biosphere and the Reasonably Maximally Exposed Individual

Radionuclides reaching the accessible environment enter the biosphere. The biosphere is the environment that the reasonably maximally exposed individual inhabits. Characteristics of the biosphere and the reasonably maximally exposed individual are based on current human behavior and environmental conditions in the Yucca Mountain region.

Ground water transporting released radionuclides to the biosphere may be used for drinking and agricultural purposes typical of current Amargosa Valley practices. Radionuclides entering the biosphere via ground water may reach the reasonably maximally exposed individual through some combination of three likely pathways: direct exposure from surface or suspended contamination; inhalation of suspended dust that has been contaminated; or ingestion of contaminated water, plants, or animal products. In the igneous activity release scenario, radionuclides are assumed to transport through the air and over the land surface by remobilization processes to the reasonably maximally exposed individual location. Radionuclides entering the biosphere from an igneous event may reach the reasonably maximally exposed individual through pathways such as direct exposure from surface or suspended contamination, inhalation of suspended dust that has been contaminated, or ingestion of contaminated plants or animal products. Dose conversion factors from Federal guidance are used to convert exposures from contaminated materials to doses for the reasonably maximally exposed individual.

4.2.7 Igneous Activity

Basaltic igneous activity has occurred for over 10 million years throughout the Yucca Mountain region. The probability of future igneous activity occurring directly at the proposed repository site is presently estimated at between 10^{-7} and 10^{-8} per year; however, the discovery of additional buried anomalies thought to represent basalt could change this value. Igneous activity can affect the repository through direct release or indirect release of radionuclides during extrusive or intrusive events, respectively. Although the likelihood of future igneous activity is very small, the potential radiological doses are large enough to make a significant contribution to postclosure risk in current performance calculations.

If rising magma intersects repository drifts, the magma could flow into drift openings (intrusive event) and possibly continue an upward ascent to the surface (extrusive event). During the extrusive phase of an igneous event, magma reaches the surface and forms a volcanic eruption. Generally, a magma conduit to the surface gradually widens during an eruption. If this were to occur at Yucca Mountain, waste packages intersected by flowing magma in the conduit could be expected to break apart and allow the erupting magma to entrain radionuclides. These radionuclides would be transported downwind in the volcanic plume and deposited on the ground surface. Through time, wind and water could erode and redeposit this possibly contaminated ash. Potential radiological dose from extrusive igneous events results primarily from inhalation of contaminated ash.

During the intrusive phase of an igneous event, rising magma could flow into open or partially backfilled drifts in response to the pressure gradient between the confined magma and drift voids. The thermal, mechanical, and chemical environment in magma would likely damage the waste packages and may alter the high-level waste form. After the magma cools, radionuclides would then be available for potential release from damaged waste packages through the ground water pathway.

4.3 Baseline of Risk Insights

This section discusses the risk insights that have been identified to date by the U.S. Nuclear Regulatory Commission (NRC) staff, related to performance of the potential repository system during the postclosure regulatory period.

The risk insights presented in this section are organized by 14 performance assessment model abstractions, also referred to as integrated subissues (Figure 4-3). This organizational format has also been used in two other primary NRC documents related to the high-level waste program, NUREG–1804 (NRC, 2003) and NUREG–1762 (NRC, 2002).

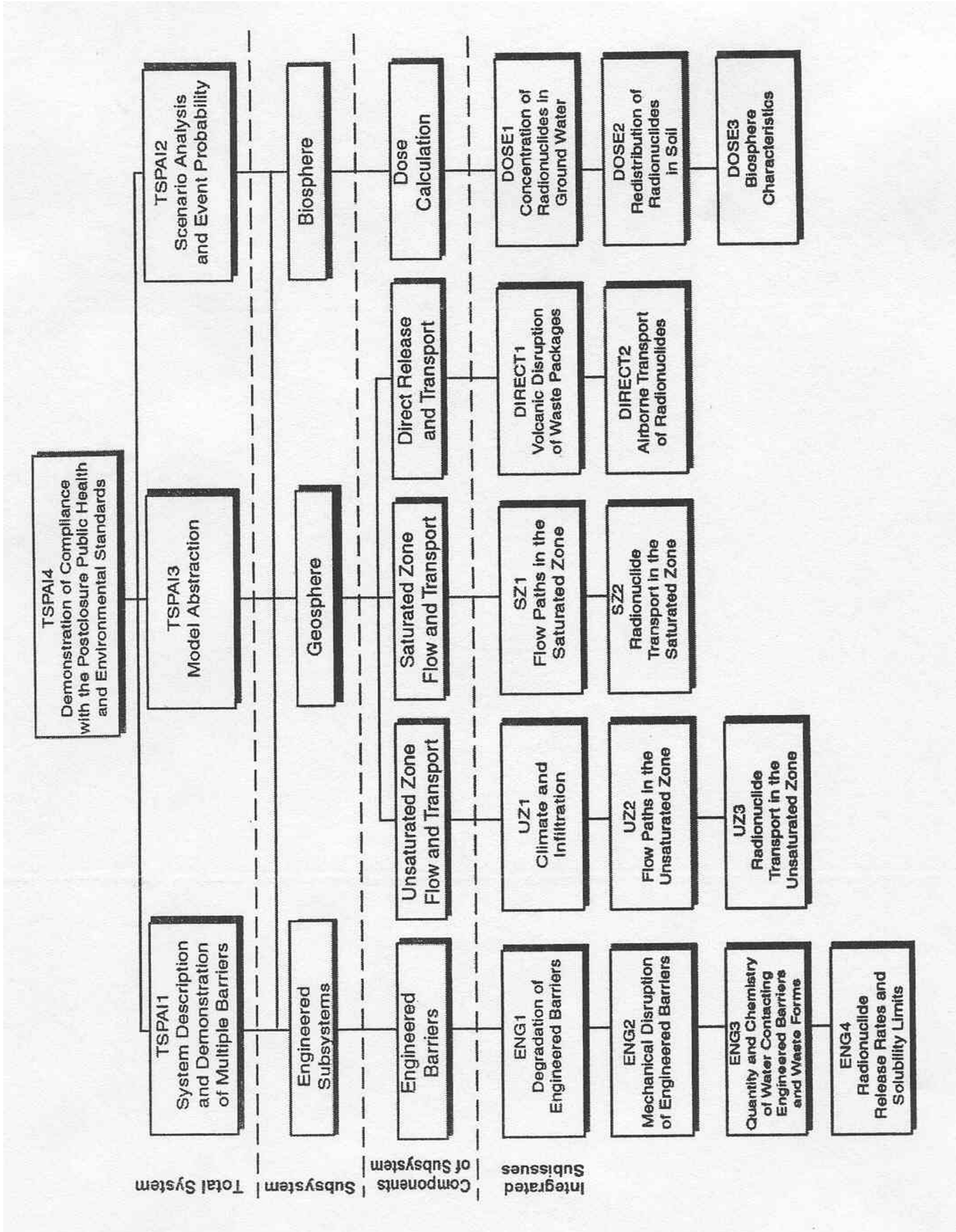

Figure 4-3. Components of Performance Assessment Review (NRC, 2003, Figure A1-5)

For each risk insight, this section provides a short title for the insight as well as a longer, more descriptive statement of the technical issue addressed by the insight. A ranking of the significance of the insight to waste isolation is provided. These rankings, which are based on risk insights, provide a transparent view of the NRC current understanding of features, events, and processes associated with a potential repository at Yucca Mountain. Such a representation of the risk insights benefits the NRC high-level waste program by providing: (i) NRC staff with information to risk-inform its review of a potential DOE license application, and (ii) other stakeholders (e.g., State of Nevada, DOE, Advisory Committee on Nuclear Waste) with information about the focus of the NRC interactions with DOE and review of a potential license application. For each risk insight, this section also provides a discussion of the technical basis for the insight, focusing as much as possible on supporting quantitative analyses as well as associated uncertainties.

Table 4-3 provides a summary of the risk insights, organized by the 14 integrated subissues, along with their significance rankings. They are presented in the table in the order in which they are presented in the following sections.

4.3.1 Degradation of Engineered Barriers (ENG1)

Risk Insights:	
Persistence of a Passive Film	High Significance
Waste Package Failure Mode	Medium Significance
Drip Shield Integrity	Medium Significance
Stress Corrosion Cracking	Medium Significance
Juvenile Failures of the Waste Package	Low Significance

4.3.1.1 Discussion of the Risk Insights

Persistence of a Passive Film: High Significance to Waste Isolation

The persistence of a passive film on the surface of the waste package is anticipated to result in very low corrosion rates of the waste package. High temperatures and aggressive water chemistry conditions have a potentially detrimental effect on the stability of the passive film and may accelerate corrosion over extended surface areas.

<u>Discussion</u>

Under environmental conditions where a stable oxide film is maintained, corrosion is uniform and occurs at a slow rate. Typical values for the passive corrosion rate of Titanium Grade 7 and Alloy 22 are in the range of 10^{-5} to 10^{-3} mm/yr [10^{-7} to 10^{-5} in/yr] (Brossia, et al., 2001; Pensado, et al., 2002). Passive corrosion rates are generally independent of pH, redox potential, and solution composition, but exhibit an Arrhenius dependence on temperature (e.g., faster corrosion rates at higher temperatures).